William J. Cahill, Jr.
Vice President

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November 14, 1975

RE: Indian Point Unit No. 2
Docket No. 50-247

Mr. Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Reactor Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Reid:

Your letter of October 14, 1975 requested information regarding the reactor vessel support system for Indian Point Unit No. 2. Attached are Con Edison's answers to Questions 1 and 2 forwarded by your Mechanical Engineering Branch.

A schedule for answering the additional questions concerning a break analysis at the reactor vessel nozzle and a reactor cavity multi-node pressure response analysis will be provided by December 8, 1975.

Very truly yours,

William J. Cahill

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Vice President

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MECHANICAL ENGINEERING BRANCH

REQUEST FOR ADDITIONAL INFORMATION

1. Provide engineering drawings of the reactor support system sufficient to show the geometry of all principal elements and materials of construction.

The reactor vessel supports consist of a box section ring girder attached to the concrete shield wall. The bottom flange and the outside circumference of the ring girder is in continuous contact with the reinforced concrete foundation (except for neutron detector openings). The support of the vessel itself is by means of four support shoes with cooling plates fastened to the ring girder. The support shoes surround a weld build-up on four alternate reactor vessel nozzles and thus permit radial thermal growth of the reactor vessel, but restrain the vessel against lateral, torsional and downward vertical motion.

The principal members of the reactor support system are shown in the drawings listed below:

9321-F-1152	Containment Interior Concrete - Reinforcing Details - Sht. 2
9321-F-1284	Reactor Vessel Support Steel
9321-F-1327	Reactor Support - Anchor Bolt Details
9321-F-1328	Metal Forms for Reactor Vessel, Sht. 1
9321-F-1329	Metal Forms for Reactor Vessel, Sht. 2
9321-F-1330	Metal Forms for Reactor Vessel, Sht. 3
9321-F 1444	Interior Concrete-Reinforcing Details
9321-F-1125	Containment Reactor Pit-Concrete and Reinforcing Details
9321-F-1126	Containment - Interior Concrete
9321-F-1151	Containment - Interior Concrete - Reinforcing Details - Sht. 1
685-J-088	Reactor Vessel - Support Hardware

Twelve copies of these drawings are attached for your use.

MECHANICAL ENGINEERING BRANCH

REQUEST FOR ADDITIONAL INFORMATION

2. Specify the detail design loads used in the original design analyses of the reactor supports giving magnitude, direction of application and the basis for each load. Also provide the calculated maximum stress in each principal element of the support system and the corresponding allowable stresses.

The basis for design of the Indian Point Unit No. 2 reactor vessel support system included consideration of breaks outside the primary shield wall. Breaks between the reactor coolant pump and the ell located before the reacto nozzle were considered for the cold legs of the reactor coolant system. On the hot legs, breaks at the steam generator inlets were analyzed.

The reactor vessel support system was sized using maximum combined load set cases of (1) 1,109 kips tangential, 322 kips radial and 1,329 kips vertical: (2) 613 kips tangential, 322 kips radial and 1,459 kips vertical: and (3) 1,006 kips tangential, 322 kips radial and 934 kips vertical. Each load set was applied to an individual support in the design phase. The loads applied in the analysis of the support system were derived from the criteria in the FSAR for Indian Point Unit No. 2.

The reactor vessel support system was analyzed for the loading combination of the normal loads, seismic loads, and the loads due to postulated pipe ruptures. These loads are considered to be statically applied to the vessel. The pipe rupture loading effect on the support considers the jet thrust acting normal to the plane of the postulated pipe rupture with a magnitude equal to the operating pressure multiplied by the pipe flow area.

Table 1 shows the maximum loads and combined loads derived from the application of this criteria to the reactor vessel supports. These loads are applied in the analysis to each individual support. Since the applied lateral loads on the reactor vessel are resisted by two support shoes, the total lateral load applied to the reactor vessel is twice that applied to each shoe. The highest lateral load the reactor vessel will see given the above described design basis is 2,218 kips. The steam generator supports and the reactor coolant pump supports attached to the unbroken loops also resist the applied lateral loads. This effect which was conservatively not included, is significant as it will reduce the loads transferred to the vessel support, since the total loads would be redistributed to all the supports.

The Indian Point Unit No. 2 Reactor Coolant System was not designed for the combination of the seismic and blowdown loads. However, an analysis for this combination of loadings has been performed for the Indian Point Unit No. 3 Reactor Coolant System, which has the same type of supports as Indian Point 2. The analysis has been performed as outlined below:

(1) A lumped mass dynamic mathematical model of the primary coolant loop and support system was developed.

(2) This dynamic model was subjected to multiple simultaneous time history hydraulic forcing functions for the blowdown analysis. The double-ended ruptures were located at places of large change in flexibility. Time history response of the total structure to these conditions was computed and reduced to time history stresses.

(3) The dynamic model was then subjected to a floor response spectra earthquake analysis.

(4) The loads as determined above were used for an evaluation of the stresses along the piping system.

(5) The stresses as determined from the basis described above are lower than the allowable stresses calculated by using the approach described in WCAP 5890, Rev. 1 and the following parameters:

(i) 20% of the uniform strain on the allowable membrane and average strain: and

(ii) 23,100 psi as the at-temperature yield in the axial direction. This value is based on the minimum value of the at-temperature yield in the loop direction as measured with samples from the Unit No. 2 piping increased by 10% for the increase in strength in-going from the loop to the axial direction. The tensile tests on the Unit No. 2 piping material at-temperature yielded at a minimum value of 20,900 psi, a maximum of 29,700 psi and an average over eleven samples of 23,300 psi.

Based on the above analysis, it is concluded that the Unit No. 2 Reactor Coolant System can stand the combination of blowdown and seismic loads within acceptable stress limits.

The concrete wall was designed to the requirements of the 1963 American Concrete Institute Code (ACI). For the applied loads in Table 1, the maximum bearing stress was determined to occur in the concrete under the ring girder. The calculated stresses at that location for combined seismic plus pipe loads were determined to be less than the ACI allowable stresses.

The ring girder was designed to the requirements of the 1963 American Institute of Steel Construction Code (AISC). For the applied loads in Table 1, the maximum stress is calculated to be less than the AISC allowable stress for non-normal loading conditions associated with plant emergency conditions. The extreme loadings of the pipe rupture are, therefore, below this stress limit used for the upset conditions.

The reactor vessel shoes were designed to the requirement of the AISC Code also. For the applied loads, the maximum stress was a shear stress in the pins between the shoe and the cooling plate/ring girder assembly. The shear stress in these pins is calculated to be less than the shear yield stress for the pins.

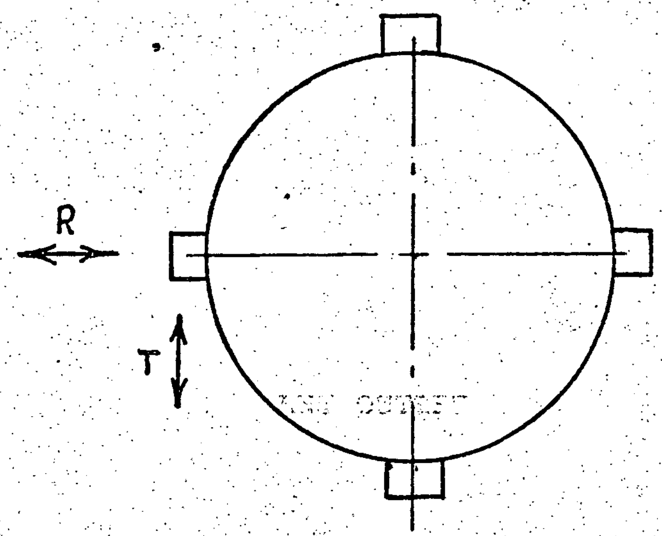
The ring girder and the shoe are both held to stresses below the yield stress of the material. This is very conservative, as additional capacity could be developed by taking the plastic response of the structures into account.

The primary shield wall was also designed for two loading conditions due to a split in the reactor. The stress in the reinforcing was limited to the tensile strength of the bars. The first load considered was a 1'0" wide longitudinal split along the length of the reactor. The vessel is assumed accelerated through a six-inch distance against the support wall by the Jet Force caused by a 2200 psi pressure which results in a live load of 625 k/ft. This load is imposed by considering an impact factor of two. The maximum rebar stress is 69.5 ksi. The second load considered a pressure build up of 1000 psi inside the pit due to release of reactor contents. This produces a rebar stress of 86 ksi. The rebar used is ASTM A 432 with an ultimate tensile strength of 90 ksi.

To protect the containment base liner an average of 2' of concrete above the containment liner plus a 1" liner plate embedded on top of the concrete was provided at the bottom of the Containment Reactor Cavity Pit. Below the containment liner plate is 4-1/2 ft. of structural concrete poured on rock.

IPP - MAXIMUM FORCES ACTING ON A REACTOR VESSEL SUPPORT

	A	B	C	D	$\sum$		
	REACTOR VESSEL WEIGHT & PIPING REACTIONS	PIPE BREAK CASE II	PIPE BREAK CASE III	EARTHQUAKE Z & VERTICAL DIRECTION	A+B	A+C	A+D
P (LB)	934,000	---	525,000	395,000	934,000	1,459,000	1,329,000
R (LB)	322,000	---	—	—	322,000	322,000	322,000
T (LB)	140,000	866,000	473,000	969,000	1,006,000	613,000	1,109,000



PLAN

PIPE BREAK CASE II IS BETWEEN PUMP AND ELL ON REACTOR NOZZLE

PIPE BREAK CASE III IS AT STEAM GENERATOR INLET

EARTHQUAKE IN Z DIRECTION ACTS NORMAL TO REACTOR COOLANT OUTLET LINE.

R - RADIAL
T - TANGENTIAL
P - VERTICAL